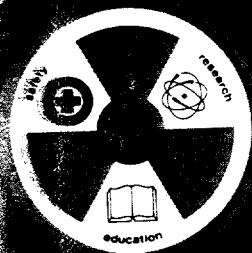


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Applied Health Physics and Safety Annual Report for 1976

OAK RIDGE NATIONAL LABORATORY
OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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HEALTH PHYSICS DIVISION

APPLIED HEALTH PHYSICS AND SAFETY ANNUAL REPORT FOR 1976

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Date Published: August 1977

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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FOREWORD

This report describes and summarizes the activities of the applied sections and/or groups of the Health Physics Division. Projects and activities within the research sections are described in ORNL-5046, Health Physics Division Annual Progress Report, Period Ending June 30, 1976.

Information in this report was contributed by, and/or compiled by the following staff members of the Applied Health Physics and Safety Sections or Groups:

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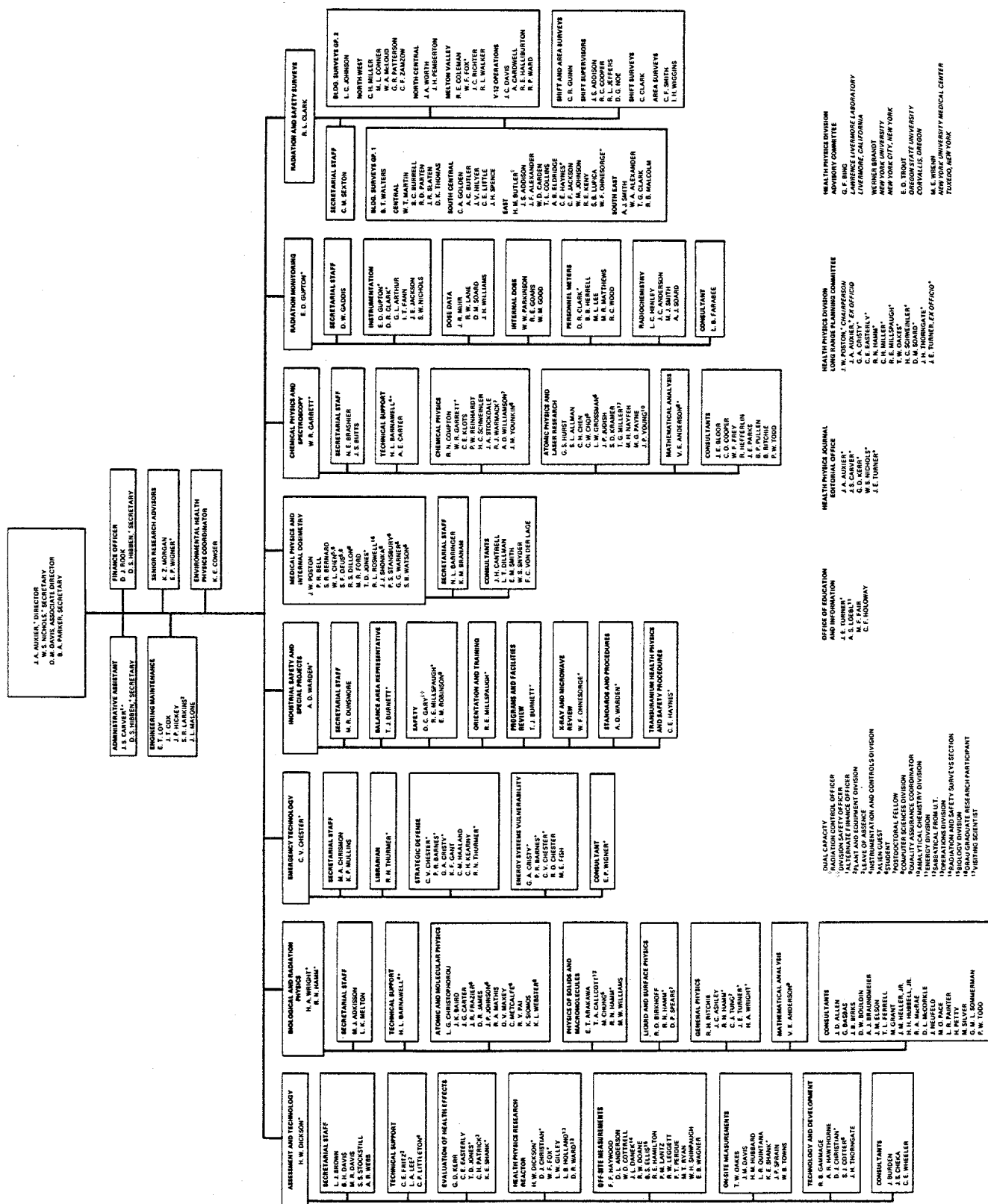
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JANUARY 1, 1977



2.0 SUMMARY

RADIATION MONITORING

Personnel Monitoring

There were no external or internal exposures to personnel which exceeded the standards for radiation protection as defined in ERDA Manual Chapter 0524. Only 61 employees received whole body radiation exposures of 1 rem or greater. The highest whole body exposure dose equivalent to an employee was 3.5 rem. The highest internal exposure was less than one-half of the maximum permissible body burden.

Health Physics Instrumentation

During 1976, 13 portable instruments were added to the inventory and 11 retired. The total number in service on January 1, 1977, was 1,284. There were seven facility radiation monitoring instruments installed and seven retired during 1976. The total number in service on January 1, 1977, was 979.

ENVIRONS MONITORING

Atmospheric Monitoring

There were no releases of gaseous waste from the Laboratory which were of a level that required an incident report to ERDA. The average concentration of beta radioactivity in the atmosphere at the perimeter of the ERDA-controlled area was less than one percent of the value applicable to releases to uncontrolled areas.

Water Monitoring

There were no releases of liquid waste from the Laboratory which were of a level that required an incident report to ERDA. The quantity of radionuclides of primary concern in the Clinch River averaged 0.51 percent of the MPC_w.

Radiation Background Measurements

The average background level at the PAM stations during 1976 was 8.6 μ R/hr, or 75 mrem/yr.

Soil Samples

Seventeen soil samples were collected and analyzed for plutonium and uranium. Plutonium-239 content ranged from 1.6×10^{-8} μ Ci/g to 4.1×10^{-8} μ Ci/g, and the uranium content ranged from 55×10^{-8} μ Ci/g to 165×10^{-8} μ Ci/g.

RADIATION AND SAFETY SURVEYS

Laboratory Operations Monitoring

During 1976, the Radiation and Safety Surveys personnel continued to assist the operating groups in keeping the contamination, air concentration, and personnel exposure levels below the established maximum permissible levels. They assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory.

Radiation Incidents

Seventeen radiation incidents involving radioactive materials were recorded during 1976. Radiation Incidents has replaced the term Unusual Occurrences and has been redefined (See Section 5.0 Radiation and Safety Surveys).

Laundry Monitoring

Of the 537,000 articles of wearing apparel monitored during 1976, about five percent were found contaminated.

INDUSTRIAL SAFETY AND SPECIAL PROJECTS

Accident Analysis

There was only one Disabling Injury experienced at ORNL in 1976, a frequency rate of 0.13. The frequency rate for 1975 was 0.27. The Serious Injury frequency rate for 1976 was 1.33, as based on the new OSHA system for recording injuries and illness (RII). The frequency rate for 1975 was 2.25.

Summary of Disabling Injuries

A total of 106 days were lost or charged for the one Disabling Injury. The employee will probably suffer a permanent-partial disability.

Safety Awards

The National Safety Council Award of Honor and the Union Carbide Corporation Award of Distinguished Safety Performance were earned by the Laboratory in 1976.

3.0 RADIATION MONITORING

3.1 Personnel Monitoring

All persons who enter Laboratory areas where there is a likelihood of exposure to radiation or radioactive materials are monitored for the kinds of exposure they are likely to sustain. External radiation dosimetry is accomplished mainly by means of badge-meters, pocket ion chambers, and hand exposure meters. Internal deposition is determined from bioassays and in vivo counting.

3.1.1 Dose Analysis Summary, 1976

(a) External Exposures - No employee received a whole body radiation dose which exceeded the standards for radiation protection, ERDA Manual Chapter 0524. The maximum whole body dose sustained by an employee was about 3.5 rem or 70 percent of the applicable standard. The range of doses to persons using ORNL badge-meters is shown in Table 3.1.1, page 10.

As of December 31, 1976, no employee had a cumulative whole body dose which was greater than the applicable standard based on the age proration formula $5(N-18)$ (Table 3.1.2, page 10). No employee has an average annual dose that exceeded five rem per year of employment (Table 3.1.3, page 10). The greatest cumulative dose of whole body radiation received by an employee was approximately 107 rem. This dose was accrued over an employment period of about 32 years and represents an average dose of about 3.3 rem per year.

The greatest cumulative dose to the skin of the whole body received by an employee during 1976 was about 16 rem or 106 percent of the applicable standard.

The maximum cumulative hand dose recorded during 1976 was about 15 rem or 20 percent of the applicable standard.

The average of the 10 greatest whole body doses to ORNL employees for each of the years 1972 through 1976 is shown in Table 3.1.4, page 11. The maximum individual dose for each of those years is shown also.

(b) Internal Exposures - Several employees were exposed to airborne material which contained ^{244}Cm . None sustained an intake as large as the minimum detectable (25% MPLB) with the Whole Body Counter. The highest exposure resulted in a systemic burden of less than 1 nCi. In another case, an employee's skin was contaminated with a dilute solution of $^{244}\text{Cm}(\text{NO}_3)_3$. The resultant systemic burden was less than 2 nCi.

There were no cases of internal exposure during the year for which the radioactive material within the body averaged as much as one-half the maximum permissible organ burden for the year.

3.1.2 External Dose Techniques

(a) Badge-Meters - Photobadge meters are issued to all employees and to nonemployees who are authorized to have frequent access to ORNL facilities. Temporary meters are issued to casual visitors.

All badge-meters are equipped with nuclear accident metering devices and beta-gamma sensitive films. Various complements of TLD's, according to potential for radiation exposure, are included in photobadge meters. NTA films are included, also, in the badges of those who are likely to be exposed to fast neutrons.

Badge-meters of employees are routinely exchanged and processed each calendar quarter, or more frequently if required for exposure control. Meters issued to visitors are processed as may be required for monitoring purposes.

(b) Pocket Meters - Pocket meters (indirect reading, ionization chambers) are made available at all principal points of entry to ORNL premises. A pair of pocket meters is carried for the duration of a work shift by persons who work in an area where the potential for an exposure of 20 mR or more exists during the work shift. Pocket meter pairs are processed each day by Health Physics technicians, and readings of 20 mR or more are reported daily to supervision. Pocket meter readings are used for estimating integrated exposure and as a basis for badge-meter processing during a calendar quarter.

(c) Hand Exposure Meters - Hand exposure meters are TLD-loaded finger rings used to measure hand dose. Hand exposure meters are issued to persons for use during operations where it is likely that the hand dose may exceed 1 rem during the week. They are issued and collected by Radiation and Safety Surveys personnel who determine the need for this type of monitoring and arrange for a processing schedule.

(d) Metering Résumé - Shown in Table 3.1.5, page 12, are the quantities of personnel metering devices used and processed during 1976. The number of dosimeters processed is less than the number issued, because those which are issued for accident dosimetry only are not processed unless there is a likelihood of exposure.

3.1.3 Internal Dose Techniques

(a) Bioassay - Urine and fecal samples are analyzed for the purpose of making internal exposure determinations. The frequency of sampling and the type of radiochemical analysis performed are based upon each specific radioisotope and the intake potential. Because of the small quantities of radioactive material in most samples, qualitative analyses are not feasible; and only quantitative analyses for pre-determined isotopes are performed routinely.

In most cases, bioassay data require interpretation to determine the dose to the person; computer programs are used for evaluation of extensive data on urinary excretion of ^{239}Pu . An estimate of dose is made for all cases in which it appears that one-fourth of a body burden averaged over a calendar year, may be exceeded.

The analyses performed by the Applied Health Physics and Safety radiochemical lab during 1976 are summarized in Table 3.1.6, page 13.

(b) Whole Body Counter - The Whole Body Counter (an in vivo gamma spectrometer) may be used for estimating internally deposited quantities of most radionuclides which emit photons.

During calendar year 1976 there were 255 whole body, thorax and wound counts. Six actinide isotopes and ten other radionuclides were identified in employees, all except the actinides at insignificant levels ($< 15\%$ maximum permissible lung burden). In the actinide cases, follow-up counting demonstrated that the lung burden decreased below the minimum detectable level within a period of a few days and urinalysis proved that systemic uptake was insignificant.

(c) Counting Facility - The Applied Health Physics and Safety counting facility determines radioactivity content of samples submitted by the applied sections. A summary of analyses is in Table 3.1.7, page 14.

3.1.4 Reports

Routine reports of personnel monitoring data are prepared and distributed to divisional supervision and to the Applied Health Physics and Safety staff.

(a) Pocket Meter Data - A report is prepared daily of the names, ORNL division, and readings for pocket meter readings which were 20 mR or greater during the previous 24 hours.

A computer-prepared report, which includes all pocket meter data for the previous week and summary data for the calendar quarter, is published and distributed weekly.

(b) External Dosimetry Data - A computer-prepared report, which includes data of recorded skin dose and whole body dose for the previous calendar quarter and totals for the current year, is published and distributed quarterly.

(c) Bioassay Data - A computer-prepared report, which includes data of sample status and results for the previous week, is published and distributed weekly. A quarterly and an annual report of results are prepared and distributed.

(d) Whole Body Counter Data - Preliminary results of analysis are reported on a card form soon after counting is done.

A computer-prepared report, which includes data collected during the previous calendar quarters of the calendar year, is published and distributed quarterly.

3.1.5 Records

Permanent records of personnel monitoring data are maintained for each person who is assigned an ORNL photobadge meter.

3.2 Health Physics Instrumentation

The Health Physics Division shares with the Instrumentation and Controls Division the responsibility for the selection of electronic radiation monitoring instruments used in the ORNL health physics program. Normally, the Health Physics Division is responsible for determining the need for new instrument types and modifications to existing types, for specifying the health physics design requirements, and for approval of the design. The Health Physics Division is also responsible for calibrating all instruments used in the health physics program and is allocated the funds for maintenance of these instruments. Maintenance is performed or cross-ordered by the Instrumentation and Controls Division.

Non-electronic personnel monitoring devices are designed, tested, calibrated and maintained by Health Physics Division personnel.

3.2.1 Instrument Inventory

The electronic instruments used in the health physics program are divided, for convenience in servicing and calibrating, into two classes: the first class includes battery-powered portable instruments; the second class includes the stationary instruments that are AC powered. Portable instruments are assigned and issued to the Radiation and Safety Surveys complexes. Stationary instruments are the property of the ORNL division which has the monitoring responsibility in the area in which the instrument is located. Table 3.2.1, page 15, lists portable instruments assigned at the end of 1976; Table 3.2.2, page 15, lists stationary instruments at the X-10 site in use at the end of 1976.

Inventory and service summaries for health physics instruments are prepared on an IBM 360. These computer-programmed reports enable the Instruments Group to maintain a current inventory on most health physics instrument requirements.

The allocation of stationary health physics monitoring instruments at the X-10 site by division is shown in Table 3.2.3, page 16.

3.2.2 Calibration Facility

The Health Physics Division maintains a calibration facility for the calibration and maintenance of portable radiation instruments and personnel metering devices. The facility is equipped with calibration sources, remote control devices, and shop space for the use of Instrumentation and Controls Division maintenance personnel. Health Physics personnel assign, arrange for maintenance of, calibrate, provide delivery services for, and maintain inventory and servicing data of all portable health physics instruments.

Portable instruments should be serviced (1) whenever repairs are needed, (2) at least once each two months for those which have replacement-type batteries, and (3) at least once each three months for those instruments which have "permanent" (rechargeable) batteries. The number of calibrations of portable instruments for 1976 is shown in Table 3.2.4, page 17.

3.3 Developments

3.3.1 Bioassay

The urinary excretion of ^{244}Cm by six employees is being followed for the possible development of an excretion curve for this element, not presently available from the literature. Four intakes were via inhalation of soluble compounds, one was via skin contact with $\text{Cm}(\text{NO}_3)_3$ and one, a case from previous years, via a puncture wound. None of the intakes was large enough to justify chelation therapy, which would have perturbed the excretion, but all were sufficient to give reliable excretion data, at least for the first 30 days. Curve fitting is in progress and preliminary results indicate that all six cases will conform within experimental error to a single excretion function. It is hoped that the derived function will permit more accurate estimates of systemic burdens of the trivalent actinides than has been possible in the past.

3.3.2 Installation of a Computer-Based Whole Body Counting System

The ORNL Whole Body Counter has recently acquired a Nuclear Data ND 6620 computer based, data acquisition and processing system. This system has a non-dedicated, high speed, decentralized synchronous bus which permits direct, bidirectional data communication between all components of the ND 6620 system. With the distributed processing techniques used here, each major device has its own microprocessor (DEC LSI 11) with firmware instructions comparable to those used in the DEC PDP 11/35 system. The system CPU (central processing unit) therefore has to send only a few simple commands to direct each controller which

then performs many complex functions without further attention from the CPU. An increase in system throughput is achieved by relieving the CPU of all the detail work that is performed by the intelligent microprocessors. Interleaved access time to system memory is 160 nsec resulting in a maximum system throughput of 6.5 megawords/sec on write transfer or 3.25 megawords/sec on read transfer.

Specifically, the ORNL system consists of a LSI 11 microcomputer-based display and acquisition system and a LSI 11 controlled CPU with 128 kilobyte memory (64 kwords). Peripherals include a dual high density disk drive unit (1 fixed, 1 removable platter), a 9-track, 25 ips, 800 bpi magnetic tape transport unit and a fast-line printer. We expect that our facility will be much more efficient in data analysis and file maintenance because of this addition. Integration of the unit with out various detector systems is now being undertaken.

3.3.3 Continued Development of Improved Techniques for In Vivo Detection of the Actinide Elements

A major effort is still being undertaken by the WBC (Whole Body Counter) staff to analyze the suitability of current radionuclide analysis techniques and to develop improved analytical and experimental methods for in vivo detection of the actinide elements, particularly ^{238}Pu , ^{239}Pu and ^{244}Cm . These research efforts may be summarized as follows:

(a) Monte Carlo simulation of the photon exit spectrum arising from a given distribution of ^{239}Pu in the lungs (with G. G. Warner of the Medical Physics and Internal Dosimetry Section and C. E. Bemis, Jr. of the Chemistry Division). These data are being used to investigate the effect of isotope distribution on WBC calibration factors and on detector placement.

(b) Continued investigation of the use of large, intrinsic Ge arrays for in vivo analysis of Pu (with C. E. Bemis, Jr.). This effort has involved the most extensive experimental and theoretical studies carried out in any laboratory to date and certain conclusions have now been reached. Monte Carlo simulation studies of L x-ray photon transport in a mathematical phantom, along with experimental studies, lead us to believe that a seven-element intrinsic germanium array (each element 5 cm² in area x 8 mm deep) would yield approximately a factor of three higher sensitivity for actinides than presently achievable with the 5"-diameter NaI phoswich system in use at the Whole Body Counter. Comparison of experiment with theory has been good and these results will be published very soon.

(c) Error analysis. Mathematical studies are being conducted to derive confidence limits for data obtained from our various counting systems.

(d) High resolution ultrasonic techniques. High resolution ultrasonic pulse-echo techniques (resolution in soft tissue ≤ 0.1 mm) have recently been developed at the Whole Body Counter. Feasibility studies are ongoing to determine the suitability of these methods for detecting small foreign objects in radioactively contaminated wounds.

Table 3.1.1 Dose Data Summary for Laboratory Population
Involving Exposure to Whole Body Radiation--1976

<u>Group</u>	<u>Number of Rem Doses in Each Range</u>							<u>Total</u>
	0-0.1	0.1-1	1-2	2-3	3-4	4-5	5 up	
ORNL Employees	5337	381	61	9	3	0	0	5791
ORNL-Monitored Non-Employees	101	9	0	0	0	0	0	110
TOTAL	5438	390	61	9	3	0	0	5901

Table 3.1.2 Average Rem Per Year Since Age 18--1976

<u>Group</u>	<u>Number of Doses in Each Range</u>				<u>Total</u>
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5786	5	0	0	5791

Table 3.1.3 Average Rem Per Year of Employment at ORNL--1976

<u>Group</u>	<u>Number of Doses in Each Range</u>				<u>Total</u>
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	
ORNL Employees	5781	10	0	0	5791

Table 3.1.4 Average of the Ten Highest Whole Body
Doses and the Highest Individual Dose by Year

Year	Average of the Ten Highest Doses (Rem)	The Highest Dose (Rem)
1972	4.18	4.88
1973	3.12	4.63
1974	2.34	3.58
1975	2.41	2.71
1976	2.68	3.49

Table 3.1.5 Personnel Meters Services

	<u>1974</u>	<u>1975</u>	<u>1976</u>
A. Pocket Meter Usage			
1. Number of Pairs Used			
ORNL	84,864	80,860	77,272
CPFF	<u>10,452</u>	<u>9,984</u>	<u>8,944</u>
Total	95,316	90,844	86,216
2. Average Number of Users per Quarter			
ORNL	804	806	747
CPFF	<u>160</u>	<u>146</u>	<u>194</u>
Total	964	952	941
B. Meters Processed for Monitoring Data			
1. Beta-Gamma Badge-Meter	18,490	23,600	20,190
2. Neutron Badge-Meter	550	680	790
3. Hand Meter	810	670	550

Table 3.1.6 Radiochemical Lab Analyses--1976

Radionuclide	Urine	Feces	Milk	Water	Controls
Plutonium, Alpha	580	9		84	60
Transplutonium Alpha	541	5		84	60
Uranium, Alpha	257	7			10
Strontium, Beta	218		502	43	42
Cesium-137	34				10
Tritium	160			180	35
Iodine-131			502		84
Other	136				5
TOTALS	1926	21	1004	391	306

Table 3.1.7 Counting Facility Analyses--1976

Types of Samples	Number of Samples			Unit Total
	Alpha	Beta	Gamma	
Facility Monitoring				
Smears	35,670	36,500		72,170
Air Filters	14,965	12,877		27,842
Environs Monitoring				
Air Filters	3,250	3,250	250	6,750
Fallout		3,142	100	3,242
Rainwater		746		746
Surface Water		303		303
Milk		468	56	524

Table 3.2.1 Portable Instrument Inventory--1976

Instrument Type	Instruments Added 1976	Instruments Retired 1976	In Service Jan. 1, 1977
G-M Survey Meter	8	8	450
Cutie Pie	3	2	423
Alpha Survey Meter	1	1	280
Neutron Survey Meter	1	0	109
Miscellaneous	0	0	22
TOTAL	13	11	1284

Table 3.2.2 Inventory of Facility Radiation Monitoring
Instruments for the Year--1976

Instrument Type	Installed During 1976	Retired During 1976	Total Jan. 1, 1977
Air Monitor, Alpha	1	2	104
Air Monitor, Beta	0	1	168
Lab Monitor, Alpha	1	0	177
Lab Monitor, Beta	2	0	214
Monitron	0	1	206
Other	3	3	110
TOTAL	7	7	979

Table 3.2.3 Health Physics Facility Monitoring Instruments
Divisional Allocation at X-10 Site--1976

ORNL Division	α Air Monitor	β Air Monitor	α Lab Monitor	β Lab Monitor	Monitron	Other	Total
Analytical Chemistry	7	12	16	18	14	5	72
Chemical Technology	51	52	72	52	55	34	316
Chemistry	8	5	14	13	3	2	45
Metals and Ceramics	11	6	14	4	5	10	50
Operations	15	79	37	80	108	32	351
All Others	12	14	24	47	21	27	145
TOTAL	104	168	177	214	206	110	979

Table 3.2.4 Calibrations Facility Résumé--1976

	1976
Beta-Gamma	2,809
Neutron	346
Alpha	921
Personal Dosimeters	5,820
Badge Dosimetry Components	11,200

4.0 ENVIRONMENTAL MONITORING

The Health Physics Division monitors for airborne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network* consists of 23 stations that are positioned relatively close to ORNL operational activities; the perimeter air monitoring (PAM) network consists of nine stations located on the perimeter of the ERDA-controlled area and provides data for evaluating the impact of all Oak Ridge operations on the immediate environment; and the remote air monitoring (RAM) network consists of eight stations located outside the ERDA-controlled area at distances of from 12 to 75 miles from ORNL. The monitoring networks provide for the collection of (1) airborne radioactivity by air filtration techniques, (2) radioparticulate fallout material by impingement on gummed paper trays, (3) rainwater for measurement of fallout occurring as rainout, and (4) radioiodine using charcoal cartridges.

Low-level radioactive liquid wastes originating from ORNL operations are discharged, after treatment, to White Oak Creek, which is a small tributary of the Clinch River. The radioactive content of White Oak Creek discharge is determined at White Oak Dam, which is the last control point along the stream prior to the entry of White Oak Creek into the Clinch River. Water samples are collected at several locations in the Clinch River, beginning at a point above the entry of the wastes into the river and ending at Center's Ferry near Kingston, Tennessee, the nearest population center downstream.

Samples of White Oak Creek effluent are collected at White Oak Dam by a continuous proportional sampler and analyzed weekly for gross beta activity as a control measure and as a means of evaluating the gross concentration of radioactivity entering the Clinch River. Portions of the weekly samples are composited into monthly samples for detailed analyses by gamma spectrometric and wet-chemical techniques. The weekly samples are analyzed for transuranic alpha emitters, total strontium, tritium, and iodine-131. The monthly composites are concentrated and analyzed by radiochemical and gamma spectrometric techniques, normally for the following: strontium-90, cesium-137, barium-140, cerium-144, ruthenium-106, zirconium-95, niobium-95, cobalt-60, tritium, plutonium, transplutonium, and gross beta. Calculations are made of the concentrations of radioactivity in the Clinch River at the point of entry of White Oak Creek, using the concentrations measured at White Oak Dam and the dilution provided by the river. To verify the calculated concentrations, two sampling stations are maintained in the Clinch River below the point of entry of the wastes; one at the Oak Ridge Gaseous Diffusion Plant (ORGDP) water intake (Clinch River Mile [CRM] 14.5) and the other

* See figures in ORNL-5169 (LAM No. 23, added in 1976, located at Walker Branch).

at Center's Ferry near Kingston, Tennessee (CRM 4.5). Additional sampling stations are maintained in the Clinch River above the point of entry of the waste at Melton Hill Dam (CRM 23.1) to provide baseline data and at the mouth of White Oak Creek for backup measurements of White Oak Dam station.

The ORGDP water sampling station collects a sample from the Clinch River proportional to the flow in the river near the water intake of the ORGDP water system. The samples are brought into the Laboratory at weekly intervals, and an aliquot is composited for quarterly analysis of tritium. The remaining portion of the sample is passed over anion and cation resins to remove nuclides. At quarterly intervals, the resin columns are eluted, and the eluent is analyzed for gross activity and for individual radionuclides that may be present in significant amounts.

A "grab" sample is collected daily at the Center's Ferry sampling station which is located on the Clinch River at CRM 4.5. The daily grab samples are composited and analyzed on a quarterly basis. The preparation of these samples and the analyses performed are the same as those for the ORGDP water sampling station.

The Melton Hill Dam sampling station collects a sample proportional to the flow of water through the power-generating turbines, which represents all of the discharge from the Dam other than a minor amount discharged in the operation of the locks. Samples are collected from the station at weekly intervals, processed, and analyzed in the same manner as for the ORGDP water sampling station.

Samples of ORNL potable water are collected daily, composited, and stored. At the end of each quarter, these composites are analyzed radiochemically for ^{90}Sr content and are assayed for long-lived gamma-emitting radionuclides by gamma spectrometry.

Raw milk is collected at 10 sampling stations located within a radius of 50 miles from ORNL. Samples are taken on a weekly basis from six stations located outside the ERDA-controlled area within a 20-mile radius of ORNL. Samples are collected every five weeks from the four remaining stations located more remotely with respect to Oak Ridge operations out to distances of about 50 miles. The purpose of the milk sampling program is twofold: first, samples collected in the immediate vicinity of ORNL provide data by which one may evaluate the possible effect of effluents from ORNL operations; second, samples collected remote to the immediate vicinity of ORNL provide background data which are essential in establishing a proper index from which releases of radioactive materials originating from Oak Ridge operations may be evaluated. The milk samples are analyzed by radiochemical techniques for strontium-90 and iodine-131. The minimum detectable concentrations of strontium-90 and iodine-131 in milk are 0.5 pCi/liter and 0.45 pCi/liter, respectively.

External gamma radiation background measurements are made routinely at the local and perimeter air monitoring stations, at one station

located near Melton Hill Dam and at the remote monitoring stations; measurements are made using calcium fluoride thermoluminescent dosimeters suspended one meter above the ground. Dosimeters at the perimeter stations and Melton Hill Dam are collected and analyzed monthly. Those at local and remote stations are collected and analyzed semiannually.

External gamma radiation measurements are also made routinely along the bank of the Clinch River from the mouth of White Oak Creek several hundred yards downstream. These measurements were used to evaluate gamma radiation levels resulting from ORNL liquid effluent releases and "sky shine" from an experimental ^{137}Cs plot located near the river bank. Radiation measurements were made using lithium fluoride thermoluminescent dosimeters suspended one meter above the ground surface.

Various species of fish, which are commonly caught and eaten, in eastern Tennessee, are taken from the Clinch River during the spring and summer of each year. Ten fish of each species are composited for each sample, and the samples are analyzed by gamma spectrometric and radiochemical techniques for the critical radionuclides which may contribute significantly to the potential radiation dose to man.

Soil samples are collected annually from locations near the PAM and RAM stations. Nine samples, approximately three inches in diameter and one centimeter thick, are collected in a one-square-meter area at each location, composited, and analyzed by gamma spectroscopy and radiochemical techniques for uranium, plutonium, and various other radioisotopes.

4.1 Atmospheric Monitoring

4.1.1 Air Concentrations

The average concentrations of beta radioactivity in the atmosphere, as measured with filters from the LAM, PAM, and RAM networks during 1976, were as follows:

<u>Network</u>	<u>Concentration ($\mu\text{Ci/cc}$)</u>
LAM	4.6×10^{-14}
PAM	2.9×10^{-14}
RAM	2.6×10^{-14}

The LAM network value of $4.6 \times 10^{-14} \mu\text{Ci/cc}$ is less than 0.002 percent of the MPCU_a ¹ based on occupational exposure of $3 \times 10^{-9} \mu\text{Ci/cc}$. Both the PAM and RAM network values represent < 0.03 percent of the MPCU_a of

¹ The MPCU_a is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. ERDA Manual Chapter 0524, Appendix, Annex 1, gives exposure values applicable to various mixtures of radionuclides and establishes guidelines for deriving the MPCU_a .

1×10^{-10} $\mu\text{Ci/cc}$ applicable to releases to uncontrolled areas. A tabulation of data for each station in each network is given in Table 4.1.1. The weekly values for each network are illustrated in Table 4.1.2.

The values measured for 1976 are lower than those for 1975 by more than 20% for the LAM, PAM, and RAM networks.

4.1.2 Fallout (Gummed Paper Technique)

The average activity from radioparticulate fallout measured for 1976 was lower than those for 1975 by a factor of ~ 2 for the LAM and PAM networks. However, seven times more particles per square foot were found in 1976 as compared with 1975 in the LAM network. The average activity and number of particles per square foot are shown in Table 4.1.3.

4.1.3 Rainout (Gross Analysis of Rainwater)

The average concentration of beta radioactivity in rain water collected from the three networks during 1976 was as follows:

<u>Network</u>	<u>Concentration ($\mu\text{Ci/ml}$)</u>
LAM	2.0×10^{-8}
PAM	1.8×10^{-8}
RAM	2.5×10^{-8}

The average concentrations of beta radioactivity measured for 1976 were approximately the same concentrations measured in 1975 for the three networks. The average concentration measured at each station within each network is presented in Table 4.1.4. The average concentration for each network for each week is given in Table 4.1.5.

4.1.4 Atmospheric Radioiodine (Charcoal Cartridge Technique)

Atmospheric iodine sampled at the perimeter stations averaged 0.8×10^{-14} $\mu\text{Ci/cc}$ during 1976. This average represents < 0.01 percent of the maximum permissible concentration of 1×10^{-10} $\mu\text{Ci/cc}$ applicable to inhalation of ^{131}I released to uncontrolled areas. The maximum concentration observed for one week was 3.3×10^{-14} $\mu\text{Ci/cc}$.

The average radioiodine concentration at the local stations was 2.6×10^{-14} $\mu\text{Ci/cc}$. This concentration is < 0.01 percent of the maximum permissible concentration for inhalation by occupational personnel. The maximum concentration for one week was 7.2×10^{-14} $\mu\text{Ci/cc}$.

Table 4.1.6 presents the ^{131}I weekly average concentration data for both the local area (LAM) and the perimeter area (PAM) air monitoring networks. The weekly average ^{131}I concentration in air measured by stations in the LAM and PAM networks are given in Table 4.1.7.

The results of the specific radionuclide analyses of the filters from the three networks are given in Table 4.1.8.

4.1.5 Milk Analysis

The yearly average and maximum concentrations of ^{131}I and ^{90}Sr in raw milk are given in Tables 4.1.9 and 4.1.10. If one assumes the average intake of milk per individual to be one liter per day, the concentrations of ^{131}I in milk collected near ORNL and in milk collected more remotely from ORNL are within FRC Range I, except for Station 51 which had excessively high levels due to fallout. The concentrations of ^{90}Sr in milk from both the immediate and remote environs of ORNL are also within FRC Range I.

The concentration of ^{90}Sr in milk varies with locations; part of the variation has been found to result from differences in farming methods. Pastureland that is not fertilized and is overgrazed (a not too uncommon practice in this area) apparently results in a higher than normal concentration of ^{90}Sr in milk from cows pastured on this land.

4.1.6 ORNL Stack Releases

The radionuclide releases from ORNL stacks are summarized in Table 4.1.11.

4.2 Water Monitoring

4.2.1 White Oak Lake Waters

Yearly discharges of specific radionuclides to the Clinch River, 1968 through 1976, are shown in Table 4.2.1.

The calculated average concentrations of the significant radionuclides in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the river) are presented in Table 4.2.2. The concentration did not exceed 1% of MPC_w for any month during 1976 (Table 4.2.3).

The annual average percent MPC_w of beta emitters, other than tritium in the Clinch River, 1968 through 1976, is given in Table 4.2.4. Table 4.2.5 lists the annual average percent MPC_w of tritium in the Clinch River, 1968 through 1976.

4.2.2 Clinch River Water

The measured average concentrations and the percent of MPC_w of radionuclides in the Clinch River at Melton Hill Dam (CRM 23.1), about three miles upstream, at Gallaher (CRM 14.5), about six miles downstream, and at Center's Ferry (CRM 4.5), about 16 miles downstream from the entry of White Oak Creek, are given in Table 4.2.2.

4.2.3 Potable Water

The average concentrations of ^{90}Sr in potable water at ORNL during 1976 were as follows:

<u>Quarter Number</u>	<u>Concentration of ^{90}Sr ($\mu\text{Ci/ml}$)</u>
1	5.0×10^{-11}
2	5.0×10^{-11}
3	5.0×10^{-11}
4	5.0×10^{-11}
Average for Year	5.0×10^{-11}

The average value of 5.0×10^{-11} represents < 0.1 percent of the MPC_w for drinking water applicable to individuals in the general population.

4.2.4 Radionuclides in Clinch River Fish

The results of the analysis of fish samples are tabulated in pCi/kg of wet weight (Table 4.2.6) for each radionuclide of significance. An estimate of man's intake of radionuclides from eating the fish is made by assuming an annual rate of fish consumption of 37 pounds. An estimated percentage of maximum permissible intake is calculated by assuming a maximum permissible intake of fish to be comparable to a daily intake of 2.2 liters of water containing the MPC_w of these radionuclides for a period of one year.

4.3 Radiation Background Measurements

Data on the average external gamma radiation background rates are given in Tables 4.3.1 and 4.3.2. The slight difference between the average levels in the perimeter and remote environs is considered to be within the variation in background levels normally experienced in East Tennessee which is dependent upon elevation, topography, and geological character of surrounding soil.²

The average external gamma radiation levels along the bank of the Clinch River adjacent to an experimental cesium field are given in Table 4.3.3.

4.4 Soil Samples

Data on uranium, plutonium, and other radioisotope concentrations in soil samples are given in Table 4.4.1.

² W. M. Lowder et al., *Indoor Radon Daughter and Radiation Measurements in East Tennessee and Central Florida*, HASL-TM-71-8, March, 1971.

4.5 Environmental Monitoring Samples

A listing of environmental monitoring samples processed by type sample, type of analyses, and number of samples is given in Table 4.5.1.

4.6 Calculation of Potential Radiation Dose to the Public

To determine the radiation doses resulting from gaseous discharges from ORNL, the Gaussian plume model developed by Pasquill³ and Gifford⁴ was incorporated into a computer program. The effluents were assumed to occur from a single 250-foot stack, and meteorological data collected at the ORNL site were used.

The average total body dose to an Oak Ridge resident from ORNL gaseous effluents was calculated to be .04 mrem/yr, which is < .01 percent of the allowable standard. The cumulative whole body dose to the population within a 50-mile radius of ORNL, resulting from 1976 atmospheric effluents, was calculated to be 4.3 man-rem. This dose may be compared to an estimated 80,000 man-rem to the same population from natural background radiation.

The point of maximum potential exposure to an individual on the site boundary is located along the bank of the Clinch River adjacent to an experimental cesium field and is due primarily to "sky shine" from the field. A maximum potential whole body exposure of 260 mrem/yr was calculated for this location, assuming that an individual remained at this point for 24 hours/day for the entire year. The calculated maximum potential exposure is 52 percent of the allowable standard.⁵ This is an atypical exposure location, and the probability of an exposure of the magnitude calculated is considered remote since access is only by boat.

The total body dose to a "hypothetical maximum exposed individual" at the same location was calculated, using a more realistic residence time of 240 hours/yr. The calculated dose under these conditions was 7.2 mrem/yr, which is 1.4 percent of the allowable standard and represents what is considered a probable upper limit of exposure.

4.7 Quality Assurance Program

The Environmental Monitoring Group at the Oak Ridge National Laboratory has initiated a quality assurance program to ensure that a

³ F. Pasquill, *Atmospheric Diffusion*, D. Van Nostrand Co., Ltd., London, 1962.

⁴ F. A. Gifford, Jr., *The Problem of Forecasting Dispersion in the Lower Atmosphere*, USAEC, DTI, 1962.

⁵ ERDA Manual Chapter 0524.

high degree of accuracy and reliability is maintained in its radiological surveillance activities. The program consists of techniques of quality control, document control, and corrective action procedures. The program includes the establishment of a detailed written description of all activities pertaining to the Environmental Monitoring Group. This includes operating procedures for each activity; inspection lists of operating and maintenance activities; checkoff frequency lists for all quality assurance steps, such as schedules for equipment inspection and test control; and documentation of compliance with quality assurance procedures. The program also involves participation in intralaboratory and interlaboratory sample-exchange programs, and evaluation of the adequacy of sample preparation work. Auditing and feedback procedures are used throughout the program. Finally, each staff member is given his role, responsibilities, and authority as related to quality assurance.

Table 4.1.1 Concentration of Beta Radioactivity in Air - 1976
(Filter Paper Data - Yearly Average)

Station Number	Location	Long-Lived Activity 10^{-14} $\mu\text{Ci/cc}$
<u>Laboratory Area</u>		
HP-1	S 3587	4.5
HP-2	NE 3025	4.8
HP-3	SW 1000	4.6
HP-4	W Settling Basin	3.5
HP-5	E 2506	9.5
HP-6	SW 3027	4.5
HP-7	W 7001	5.2
HP-8	Rock Quarry	4.2
HP-9	N Bethel Valley Road	4.8
HP-10	W 2075	2.8
HP-16	E 4500	3.5
HP-20	HFIR	3.5
HP-23	Walker Branch	4.2
Average		4.6
<u>Perimeter Area</u>		
HP-31	Kerr Hollow Gate	2.2
HP-32	Midway Gate	2.6
HP-33	Gallaher Gate	2.7
HP-34	White Oak Dam	2.5
HP-35	Blair Gate	4.2
HP-36	Turnpike Gate	3.6
HP-37	Hickory Creek Bend	1.8
HP-38	E EGCR	3.0
HP-39	Townsite	3.1
Average		2.9
<u>Remote Area</u>		
HP-51	Norris Dam	1.8
HP-52	Loudoun Dam	2.5
HP-53	Douglas Dam	3.4
HP-54	Cherokee Dam	4.2
HP-55	Watts Bar Dam	2.0
HP-56	Great Falls Dam	2.1
HP-57	Dale Hollow Dam	2.2
HP-58	Knoxville	2.5
Average		2.6

Table 4.1.2 Concentration of Beta Radioactivity in Air
as Determined from Filter Paper Data - 1976
(System Average - by Weeks)

Units of 10^{-14} $\mu\text{Ci/cc}$				Units of 10^{-14} $\mu\text{Ci/cc}$			
Week Number	LAM's	PAM's	RAM's	Week Number	LAM's	PAM's	RAM's
1	2.7	1.5	1.5	29	2.8	1.5	0.93
2	2.8	1.8	1.6	30	3.6	1.4	1.3
3	2.3	1.5	1.3	31	2.8	1.2	1.1
4	2.5	1.3	1.1	32	7.2	1.8	1.1
5	2.0	1.5	1.3	33	3.1	1.7	1.4
6	2.0	1.5	1.3	34	2.7	1.4	1.2
7	2.4	1.4	1.4	35	3.1	1.4	1.1
8	2.1	1.8	1.8	36	5.8	2.1	1.5
9	3.1	2.4	2.1	37	3.4	1.6	1.3
10	3.3	2.5	1.5	38	3.6	2.6	2.1
11	2.7	2.1	1.5	39	2.9	1.7	1.5
12	2.7	1.7	1.4	40	16	29	6.2
13	2.0	1.4	1.1	41	37	11	31
14	2.5	1.6	1.3	42	6.5	5.3	6.1
15	3.0	1.6	1.6	43	14	10	9.2
16	3.9	2.2	2.4	44	12	7.0	5.8
17	2.7	1.5	1.4	45	8.0	5.0	4.0
18	4.2	1.2	0.88	46	7.1	4.6	4.5
19	3.7	1.1	1.1	47	7.7	3.7	4.3
20	2.7	1.0	1.3	48	4.3	3.0	3.1
21	3.6	2.7	1.2	49	4.2	3.0	2.5
22	1.6	0.98	1.1	50	4.1	2.8	2.2
23	1.7	1.0	0.92	51	3.0	2.2	2.0
24	2.9	2.0	2.0	52	2.7	1.7	1.5
25	1.9	1.2	1.1	53	2.6	1.8	1.7
26	0.78	1.0	0.57				
27	3.2	1.6	0.78				
28	2.3	1.9	1.6	Average	4.6	2.9	2.6

Table 4.1.3 Radioparticulate Fallout - 1976
(Gummed Paper Data - Station Yearly Average)

Station Number	Location	Long-Lived Beta Activity 10^{-4} $\mu\text{Ci}/\text{ft}^2$	Total Particles Per Sq. Ft.*
<u>Laboratory Area</u>			
HP-1	S 3587	0.07	5.3
HP-2	NE 3025	0.03	9.1
HP-3	SW 1000	0.04	8.3
HP-4	W Settling Basin	0.06	11
HP-5	E 2506	0.12	21
HP-6	SW 3027	0.04	11
HP-7	W 7001	0.03	8.3
HP-8	Rock Quarry	0.03	12
HP-9	N Bethel Valley Road	0.03	6.2
HP-10	W 2075	0.05	10
HP-16	E 4500	0.03	7.2
HP-20	HFIR	0.03	6.6
HP-23	Walker Branch	0.03	7.6
Average		0.05	9.5
<u>Perimeter Area</u>			
HP-31	Kerr Hollow Gate	0.04	ND **
HP-32	Midway Gate	0.04	0.19
HP-33	Gallaher Gate	0.03	ND
HP-34	White Oak Dam	0.03	ND
HP-35	Blair Gate	0.05	0.19
HP-36	Turnpike Gate	0.03	ND
HP-37	Hickory Creek Bend	0.03	ND
HP-38	E EGCR	0.03	ND
HP-39	Townsite	0.07	0.38
Average		0.04	0.08
<u>Remote Area</u>			
HP-51	Norris Dam	0.04	ND
HP-52	Loudoun Dam	0.04	ND
HP-53	Douglas Dam	0.01	0.19
HP-54	Cherokee Dam	0.03	0.38
HP-55	Watts Bar Dam	0.06	ND
HP-56	Great Falls Dam	0.06	ND
HP-57	Dale Hollow Dam	0.04	ND
HP-58	Knoxville	0.04	ND
Average		0.03	0.07

* Data determined from autoradiograms.

** None Detected.

Table 4.1.4 Concentration of Beta Radioactivity in Rainwater - 1976
(Yearly Average by Stations)

Station Number	Location	Activity in Collected Rainwater 10^{-8} $\mu\text{Ci/ml}$
<u>Laboratory Area</u>		
HP-7	West 7001	1.9
HP-23	Walker Branch	2.1
Average		2.0
<u>Perimeter Area</u>		
HP-31	Kerr Hollow Gate	1.7
HP-32	Midway Gate	1.7
HP-33	Gallagher Gate	2.0
HP-34	White Oak Dam	1.7
HP-35	Blair Gate	1.7
HP-36	Turnpike Gate	1.9
HP-37	Hickory Creek Bend	1.2
HP-38	E EGCR	2.4
HP-39	Townsite	1.6
Average		1.8
<u>Remote Area</u>		
HP-51	Norris Dam	3.0
HP-52	Loudoun Dam	2.5
HP-53	Douglas Dam	1.7
HP-54	Cherokee Dam	3.7
HP-55	Watts Bar Dam	1.7
HP-56	Great Falls Dam	2.3
HP-57	Dale Hollow Dam	2.7
HP-58	Knoxville	2.6
Average		2.5

Table 4.1.5 Weekly Average Concentration of Beta
Radioactivity in Rainwater - 1976
(Units of 10^{-8} $\mu\text{Ci/ml}$)

Week Number	LAM's	PAM's	RAM's	Week Number	LAM's	PAM's	RAM's
1	0.45	0.77	1.8	29	0.55	0.79	0.77
2	1.3	1.2	1.5	30	1.5	0.42	1.3
3	0.90	0.76	0.97	31	0.55	0.62	1.2
4	0.40	0.38	1.02	32	0.95	0.98	0.80
5	1.55	0.74	0.49	33	0.80	0.47	1.1
6	0.60	0.76	1.46	34	*	*	*
7	*	*	2.9	35	1.1	0.44	0.85
8	0.40	0.53	1.1	36	0.90	0.91	1.5
9	*	*	*	37	*	*	0.12
10	*	0.73	0.47	38	*	0.20	0.97
11	1.0	0.44	0.92	39	0.30	0.57	0.27
12	0.75	0.88	0.95	40	1.6	0.97	0.45
13	1.2	0.73	1.1	41	8.7	7.1	6.7
14	1.1	1.2	1.2	42	*	*	29
15	*	*	1.2	43	21	16	18
16	*	*	*	44	12	13	15
17	0.75	0.56	0.93	45	*	*	*
18	0.95	0.44	0.35	46	7.8	7.2	12
19	0.10	0.76	0.87	47	*	*	*
20	0.50	0.44	0.80	48	2.7	1.8	3.9
21	0.65	*	1.0	49	1.8	1.1	2.9
22	0.05	0.21	0.57	50	1.5	1.0	1.2
23	1.4	0.71	1.2	51	1.9	1.8	2.4
24	*	*	*	52	1.7	1.0	2.4
25	0.1	0.82	0.80	53	2.3	1.4	*
26	*	0.19	0.17				
27	0.20	0.34	0.57				
28	1.7	1.5	2.8	Average	2.0	1.8	2.5

* No rainfall.

Table 4.1.6 Weekly Concentration of ^{131}I in Air - 1976
(Units of 10^{-14} $\mu\text{Ci/cc}$)

Week Number	LAM's	PAM's	Week Number	LAM's	PAM's
1	2.2	1.3	29	7.2	0.7
2	1.4	0.6	30	3.0	0.6
3	4.2	1.0	31	2.4	0.8
4	1.7	0.9	32	1.8	0.7
5	2.4	0.8	33	1.5	0.4
6	4.8	1.0	34	2.2	0.6
7	2.3	0.7	35	3.0	0.6
8	3.7	1.0	36	3.4	0.6
9	2.6	0.8	37	2.6	0.6
10	3.6	0.5	38	4.4	1.0
11	2.2	0.7	39	2.6	0.5
12	2.2	0.8	40	2.1	3.3
13	1.8	0.7	41	4.9	1.3
14	1.6	1.0	42	2.1	0.9
15	2.1	1.2	43	2.1	0.9
16	1.7	0.8	44	1.9	0.9
17	1.8	0.5	45	1.7	1.0
18	2.9	0.9	46	3.6	1.2
19	2.8	0.8	47	3.0	0.9
20	3.3	0.8	48	2.2	0.9
21	1.3	0.7	49	1.4	0.6
22	1.1	0.6	50	3.6	0.9
23	3.3	0.7	51	2.6	1.1
24	1.9	0.6	52	1.8	0.6
25	2.6	0.8	53	1.9	0.5
26	2.7	0.7			
27	2.6	0.6			
28	3.9	0.5	Average	2.6	0.8

Table 4.1.7 Concentration of ^{131}I in Air - 1976
(Weekly Average by Stations)

Station Number	Location	Activity in Air $10^{-14} \mu\text{Ci/cc}$
<u>Laboratory Area</u>		
HP-4	W Settling Basin	2.8
HP-6	SW 3027	1.9
HP-7	W 7001	3.6
HP-8	Rock Quarry	3.1
HP-9	N Bethel Valley Roak	2.1
HP-10	W 2075	2.7
HP-16	E 4500	3.7
HP-20	HFIR	2.0
HP-23	Walker Branch	1.5
Average		2.6
<u>Perimeter Area</u>		
HP-31	Kerr Hollow Gate	0.8
HP-32	Midway Gate	1.0
HP-33	Gallaher Gate	0.8
HP-34	White Oak Dam	0.8
HP-35	Blair Gate	0.8
HP-36	Turnpike Gate	0.9
HP-37	Hickory Creek Bend	0.9
HP-38	E EGCR	0.8
HP-39	Townsite	0.8
Average		0.8

Table 4.1.8 Continuous Air Monitoring Data
Specific Radionuclides in Air
(Composite Samples)
1976
Units of 10^{-15} $\mu\text{Ci/ml}$

Radionuclides	Yearly Average		
	Local Stations	Perimeter Stations	Remote Stations
^7Be	180	141	134
^{60}Co	< 3.2	< 0.13	< 0.055
^{90}Sr	< 1.9	< 0.77	< 0.55
^{95}Zr	< 11	< 5.05	< 5.7
^{95}Nb	< 15	< 7.1	< 8.05
^{103}Ru	< 27	< 19	< 18
^{106}Ru	< 0.50	< 0.88	< 0.69
^{125}Sb	< 0.18	< 0.22	< 0.18
^{137}Cs	< 1.6	< 0.55	< 0.47
^{144}Ce	< 2.9	< 2.4	< 2.5
^{228}Th	< 0.020	< 0.014	< 0.014
^{230}Th	< 0.084	< 0.024	< 0.017
^{232}Th	< 0.017	< 0.012	< 0.0088
^{234}U	< 1.5	< 2.6	< 0.12
^{235}U	< .061	< 0.10	< 0.0090
^{238}U	< 0.48	< 0.63	< 0.059
^{238}Pu	< 0.0069	< 0.0012	< 0.00082
^{239}Pu	< 0.056	< 0.0088	< 0.0066
^{140}Ba	< 27	< 7.8	< 17
^{140}La	< 18	< 13.0	< 11
^{141}Ce	< 29	< 22.0	< 23

Table 4.1.9 Concentration of ^{131}I in Raw Milk - 1976

Station Number	No. of Samples	Units of 10^{-9} $\mu\text{Ci/ml}$			Comparison with Standard ^b
		Maximum	Minimum ^a	Average	
Immediate Environs ^c					
1	46	19	< 0.45	< 1.3	FRC Range I
2	48	17	"	< 1.4	"
3	47	36	"	< 3.2	"
4	41	64	"	< 4.9	"
5	25	50	"	< 3.4	"
6	48	54	"	< 3.7	"
Average				3.0 \pm 0.9	"
Remote Environs ^d					
51	5	42	< 0.45	< 14.30	FRC Range II
52	9	14	"	< 2.43	FRC Range I
53	8	3	"	< 0.83	"
54	9	30	"	< 4.15	"
Average				5.4 \pm 2.3	"

^aMinimum Detectable Concentration of ^{131}I is 0.45×10^{-9} $\mu\text{Ci/ml}$.

^bApplicable FRC Standard, assuming 1 liter per day intake:

Range I	0 to 1×10^{-8} $\mu\text{Ci/ml}$	- Adequate surveillance required to confirm calculated intakes.
Range II	1×10^{-8} $\mu\text{Ci/ml}$ to 1×10^{-7} $\mu\text{Ci/ml}$	- Active surveillance required.
Range III	1×10^{-7} $\mu\text{Ci/ml}$ to 1×10^{-6} $\mu\text{Ci/ml}$	- Positive control action required.

Note: Upper limit of Range II can be considered the concentration guide.

^cSee Fig. 4.0.6, ORNL-5169.

^dSee Fig. 4.0.7, ORNL-5169.

Table 4.1.10 Concentration of ^{90}Sr In Raw Milk - 1976

Station Number	No. of Samples	Units of 10^{-9} $\mu\text{Ci/ml}$			Comparison with Standard ^b
		Maximum	Minimum ^a	Average	
Immediate Environs ^c					
1	45	6.8	1.8	2.9	FRC Range I
2	49	3.0	1.1	2.0	"
3	49	5.6	0.68	2.4	"
4	41	4.1	0.68	2.0	"
5	28	6.6	1.8	3.8	"
6	49	5.5	1.8	3.8	"
Average				2.8 ± 0.6	"
Remote Environs ^d					
51	5	4.1	2.7	3.5	FRC Range I
52	9	2.2	0.68	1.4	"
53	8	3.4	1.1	2.7	"
54	9	6.1	1.8	3.4	"
Average				2.8 ± 0.16	"

^aMinimum Detectable Concentration of ^{90}Sr in milk is 0.5×10^{-9} $\mu\text{Ci/ml}$.

^bApplicable FRC Standard, assuming 1 liter per day intake:

Range	I	$0 \text{ to } 2 \times 10^{-8}$ $\mu\text{Ci/ml}$	- Adequate surveillance required to confirm calculated intakes.
Range	II	2×10^{-8} $\mu\text{Ci/ml}$ to 2×10^{-7} $\mu\text{Ci/ml}$	- Active surveillance required.
Range	III	2×10^{-7} $\mu\text{Ci/ml}$ to 2×10^{-6} $\mu\text{Ci/ml}$	- Positive control action required.

Note: Upper limit of Range II can be considered the concentration guide.

^cSee Fig. 4.0.6, ORNL-5169.

^dSee Fig. 4.0.7, ORNL-5169.

Table 4.1.11 Annual Discharges of Radionuclides to the Atmosphere^a
(Curies)

Stack Number	³ H	⁸⁵ Kr	¹³¹ I	¹³³ Xe	Pu ^b	U ^b
3039	< 6,000	< 9,300	1.08	< 45,000	ND ^c	ND
7025	< 18.6	ND	ND	ND	ND	ND
7911	ND	< 2,200	0.17	< 11,000	ND	ND
Bldg. 9204-3 Stack (Y-12)					4.0 x 10 ⁻⁶	1.0 x 10 ⁻⁶
Total	< 6,019	< 11,500	1.25	< 54,000	4.0 x 10 ⁻⁶	1.0 x 10 ⁻⁶

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^aData furnished by Operations Division.

^bMixture of all isotopes.

^cNot detectable.

Table 4.2.1 Annual Discharges of Radionuclides to the Clinch River
(Curies)

Year	^{137}Cs	^{106}Ru	^{90}Sr	^{95}Zr	^{95}Nb	Trans U Alpha	^3H
1968	1.1	5.2	2.8	0.27	0.27	0.04	9700
1969	1.4	1.7	3.1	0.18	0.18	0.2	12200
1970	2.0	1.2	3.9	0.02	0.02	0.4	9500
1971	0.93	0.50	3.4	0.01	0.01	0.05	8900
1972	1.7	0.52	6.5	0.01	0.01	0.05	10600
1973	2.3	0.69	6.7	0.05	0.05	0.08	15000
1974	1.2	0.22	6.0	0.02	0.02	0.02	8600
1975	0.62	0.30	7.2	NA*	NA	0.02	11000
1976	0.24	0.16	4.5	NA	NA	0.01**	7400

8.14
Σ 4.9

* NA - No analysis performed.

** Radionuclides identified from yearly composite sample.

Table 4.2.2 Radionuclides In The Clinch River - 1976

Location	No. of Samples	Range	Concentration of Radionuclides of Primary Concern Units of 10^{-9} μ Cl/ml				% MPC
			^{90}Sr	^{137}Cs	^{106}Ru	^3H	
C-2 CRM 23.1 ^b	4	Max.	0.09	0.05	0.18	770	
		Min.	0.05	< 0.01	0.09	460	
		Avg.	0.08 \pm 0.01	< 0.02 \pm 0.01	0.13 \pm 0.02	610 \pm 65	< 0.05
CRM 20.8 ^c	12	Max.	2.6	0.20	0.08	4000	
		Min.	0.17	0.01	0.01	320	
		Avg.	1.28 \pm 0.23	0.07 \pm 0.02	0.04 \pm 0.01	2000 \pm 320	0.51
C-3 CRM 14.5 ^b	4	Max.	0.36	0.05	0.23	3500	
		Min.	0.14	0.01	0.09	590	
		Avg.	0.26 \pm 0.05	0.03 \pm 0.01	0.14 \pm 0.03	1700 \pm 650	0.15
C-5 CRM 4.5 ^b	4	Max.	0.41	0.05	0.27	2600	
		Min.	0.14	0.01	0.09	1000	
		Avg.	0.24 \pm 0.06	0.02 \pm 0.01	0.15 \pm 0.04	1900 \pm 400	0.15

^a Most restrictive concentration guide for each isotope used for calculating percent concentration guide. The method for calculating percent of concentration guide for a known mixture of radionuclides is given in ERDA Manual, Appendix 0524, Annex A.

^b Measured values in the Clinch River.

^c Values given for this location are calculated values based on the concentrations measured at White Oak Dam and the dilution afforded by the Clinch River. They do not include radioactive materials (e.g., fallout) that may enter the river upstream of White Oak Creek outfall (CRM 20.8).

Table 4.2.3 Calculated Percent MPC_w of ORNL
Radioactivity Releases in Clinch River Water Below the
Mouth of White Oak Creek - 1976

Month	% MPC _w
January	0.89
February	0.62
March	0.72
April	0.59
May	1.0
June	0.7
July	0.33
August	0.07
September	0.14
October	0.28
November	0.35
December	0.35
Average	0.51

Table 4.2.4 Annual Average Percent MPC_w of Beta Emitters,
Other than Tritium, in the Clinch River

Year	CRM 23.1 ^a	CRM 20.8 ^b	CRM 14.5 ^a	CRM 4.5 ^a
1968	0.17	0.83	0.37	0.52
1969	0.30	0.36	0.48	0.41
1970	0.22	0.27	0.53	0.47
1971	0.21	0.20	0.65	0.44
1972	0.18	0.26	0.58	0.48
1973	0.24	0.49	0.47	0.62
1974	0.06	0.36	0.26	0.21
1975	0.03	0.43	0.14	0.12
1976	0.05	0.44	0.23	0.15

^a Values given for this location are based on analyses of water taken directly from the river.

^b Values given for this location are calculated from the levels of radionuclides released from White Oak Dam and dilution provided by the Clinch River. The contribution from upstream as measured at CRM 23.1 is not included.

Table 4.2.5 Annual Average Percent MPC_w
of Tritium in the Clinch River

Year	CRM 20.8 ^a
1968	0.07
1969	0.11
1970	0.05
1971	0.04
1972	0.04
1973	0.07
1974	0.04
1975	0.06
1976	0.07

^a Values given are calculated from the level of waste released from White Oak Dam and dilution provided by the Clinch River.

Table 4.2.6 Radionuclide Content of Clinch River and Melton Hill Lake Fish - 1976
pCi/kg Wet Wt. - Flesh

Species	Location	Number of Samples ^b	Isotopes ^a					Est. % MPI ^c
			⁹⁰ Sr	⁶⁰ Co	¹³⁷ Cs	²²⁶ Ra	²³⁹ Pu	
Shad	CRM 12.0 Below Mouth of Poplar Creek	1	8	21	184	19	0.29	0.083
White Crappie	CRM 20	1	1100	67.4	3417	105	0.23	8.1
Shad	Below Mouth of	1	26	25	438	32	0.27	0.26
Bass	White Oak Creek	1	420.9	79	5155	96	0.02	3.5
Buffalo Carp	CRM 22.0	1 ^d	7	6	133	12	0.03	0.067
Shad	Below Melton Hill Dam and above Mouth of White Oak Creek	1	25	51	857	26	0.40	0.28
Crappie	CRM 32	1	2.3	7.8	28	13	0.05	0.023
Blue Gill	Above Melton	1	8.2	12	21	13	ND	0.065
Carp	Hill Dam	1	6.4	3.2	2.7	4.6	1.5	0.047
Bass		1	26	7.7	80	6.4	0.18	0.19
Bluegill	CRM 41	1	26	6.2	33	6.2	0.3	0.19
Shad	Bay close to C.A.R.L.	1	5.1	4.7	4.1	78	0.2	0.039

^a Values for other isotopes are available from Environmental Monitoring Section of Industrial Safety and Applied Health Physics Division.

^b Composite of 10 fish in each species, unless otherwise noted.

^c Maximum Permissible Intake - Intake of radionuclide from eating fish calculated to be equal to a daily intake of 2.2 liters of water, over a period of one year, containing the concentration guide of radionuclides in question. Consumption of fish is assumed to be 37 lb/yr of the species in question. Only man-made radionuclides were used in the calculation.

^d Sample is one individual, 22 lbs total wet weight.

Table 4.3.1 External Gamma Radiation Measurements
at Local Air Monitoring Stations - 1976

Station Number	$\mu\text{R/hr}$	mrem/yr ^a \pm % ^b
HP-1	115	1000 \pm 9
HP-2	98	860 \pm 11
HP-3	9.0	78.9 \pm 3
HP-4	290	2500 \pm 4
HP-5	50	449 \pm 10
HP-6	120	1100 \pm 9
HP-7	8.3	72 \pm 7
HP-8	8.4	73 \pm 1
HP-9	10	90 \pm 1
HP-10	15.3	130 \pm 5
HP-11	11	96 \pm 3
HP-12	53	47 \pm 3
HP-13	220	1900 \pm 3
HP-14	76	670 \pm 82
HP-15	11	99 \pm 3
HP-16	9.5	80 \pm 2
HP-17	13	110 \pm 1
HP-18	No TLD	
HP-19	14	130 \pm 5
HP-20	11	93 \pm 15
HP-21	9.5	83 \pm 2
HP-22	15	130 \pm 5

^a Calculated assuming that an individual remained at this point for 24 hours/day for the entire year.

^b Percentage coefficient of variation.

Table 4.3.2 External Gamma Radiation Measurements - 1976

Station Number	Location	Background	
		$\mu\text{R/hr}$	mrem/yr \pm % ^a
Perimeter Stations			
HP-31	Kerr Hollow Gate	9.2	80 \pm 10
HP-32	Midway Gate	11	96 \pm 10
HP-33	Gallaher Gate	8.0	70 \pm 14
HP-35	Blair Gate	13	116 \pm 23
HP-36	Turnpike Gate	7.4	64 \pm 12
HP-37	Hickory Creek Bend	7.6	66 \pm 10
HP-38	East of EGCR	7.5	65 \pm 10
HP-39	Townsite	7.8	68 \pm 14
HP-40	Melton Hill	5.8	50 \pm 10
Average		8.6	75 \pm 12
Remote Stations			
HP-51	Norris Dam	6.5	54 \pm 5
HP-52	Loudoun Dam	9.0	78 \pm 13
HP-53	Douglas Dam	8.1	71 \pm 12
HP-54	Cherokee Dam	8.0	70 \pm 2
HP-55	Watts Bar Dam	7.9	69 \pm 21
HP-56	Great Falls Dam	6.9	60 \pm 17
HP-57	Dale Hollow Dam	9.8	85 \pm 26
HP-58	Knoxville	12	100 \pm 23
Average		8.5	74 \pm 12

^a Percentage coefficient of variation.

Table 4.3.3 External Gamma Radiation Measurements Along
the Perimeter of the ERDA - Oak Ridge Controlled Area - 1976

Location ^a	$\mu\text{R/hr}$	$\text{mrem/yr}^b \pm \%^c$
HP-41	16.3	143 \pm 25
HP-42	24.8	217 \pm 19
HP-60	12.7	111 \pm 13
HP-61	22.5	197 \pm 36
HP-62	32.0	280 \pm 12
HP-63	55.0	482 \pm 33
HP-64	40.2	352 \pm 45
HP-65	37.8	331 \pm 6
HP-66	35.0	307 \pm 23
HP-67	20.5	180 \pm 27
HP-68	11.6	102 \pm 15
HP-69	11.4	99 \pm 14

^a See Fig. 4.0.8, ORNL-5169.

^b Calculated assuming that an individual remained at this point for 24 hours/day for the entire year.

^c Percentage coefficient of variation.

Table 4.4.1 Radioisotope Concentrations in Soil - 1976
(Units of pCi/g - Dry)^a

Sampling Location	Number of Samples ^b	Isotopes ^c					
		⁹⁰ Sr	¹³⁷ Cs	²³⁸ Pu	²³⁹ Pu	²³⁸ U	²³⁴⁻²³⁵ U
<u>Perimeter</u>							
HP-31	1	0.39	1.4	0.0027	0.021	0.86	0.56
HP-32	1	0.51	1.8	0.0041	0.026	0.41	0.68
HP-33	1	1.2	1.4	0.0018	0.021	0.31	0.40
HP-34	1	0.41	2.8	0.0045	0.031	0.28	0.41
HP-35	1	0.71	1.5	0.0018	0.021	0.47	0.59
HP-36	1	2.1	2.4	0.0023	0.041	0.38	0.55
HP-37	1	NA*	0.3	NA	NA	NA	NA
HP-38	1	NA	1.1	NA	NA	NA	NA
HP-39	1	0.65	1.7	0.0041	0.021	0.26	0.29
Average		0.85	1.6	0.0030	0.026	0.42	0.50
<u>Remote</u>							
HP-51	1	<0.45	1.1	NA	NA	0.33	0.46
HP-52	1	<4.5	1.3	<0.0018	0.021	0.61	0.66
HP-53	1	<0.59	1.3	NA	NA	0.71	0.94
HP-54	1	<1.4	1.3	0.0018	0.018	0.24	0.41
HP-55	1	<5.0	0.88	<0.0018	0.017	0.22	0.33
HP-56	1	<0.77	0.18	NA	NA	0.40	0.48
HP-57	1	<0.68	2.6	NA	NA	0.48	0.55
HP-58	1	<1.4	1.3	0.0027	0.016	0.40	0.44
Average		<1.8	1.2	<0.002	0.018	0.42	0.53

^a Applicable guides for soil contamination have not been established.

^b Nine samples, approximately three inches in diameter and one centimeter thick, collected in a one-square-meter area at each location and composited for analysis.

^c Values for other isotopes are available from Environmental Monitoring Section of Industrial Safety and Applied Health Physics Division.

* Not analyzed.

Table 4.5.1 Environmental Monitoring Samples - 1976

Sample Type	Type of Analyses	Number of Samples
Monitoring Network Air Filters	Gross Alpha, Gross Beta	1664
Monitoring Network Air Filters	Autoradiogram	1248
Monitoring Network Air Filters	Gamma Spectrometry, Wet-Chemistry	12 Groups
Gummed Paper Fallout Trays	Autoradiogram	1248
Gummed Paper Fallout Trays	Long-Lived Activity Count	1664
Charcoal Cartridge	^{131}I	1248
Fish	Radiochemical, Gamma Spectrometry	10 Groups
Rainwater	Gross Beta	938
Raw Milk	^{131}I , ^{90}Sr	468
White Oak Dam Effluent	Gross Beta, Radiochemical, Gamma Spectrometry	408
White Oak Creek	Gross Beta, Radiochemical, Gamma Spectrometry	236
Clinch River Water	Radiochemical, Gamma Spectrometry	54
Potable Water	Radiochemical, Gamma Spectrometry	20
Soil Samples	Gamma Spectrometry, Wet-Chemistry	20

5.0 RADIATION AND SAFETY SURVEYS

5.1 Laboratory Operations Monitoring

During 1976, Radiation and Safety Surveys personnel assisted the operating groups in keeping the contamination, air concentration, and personnel exposure levels well below the established maximum permissible limits. Through seminars, safety meetings, and informal discussions with supervision, they assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory. The following is a brief description of some of the activities monitored during the year.

5.1.1 Bulk Shielding Reactor, Building 3010

Additional efforts were necessary during this year to repair leaks in the walls and floor of the Bulk Shielding Reactor Pool. Monitoring and surveillance were provided for all operations related to this program: the transfer of all fuel elements and the large (2.3×10^4 Ci) ^{60}Co source to the Graphite Reactor Canal; the relocation of other radioactive equipment items to the north end of the pool, and radiation/contamination surveys of the work areas involved. Personnel exposures resulting from this work were only a small percent of the permissible levels. Contamination controls imposed were entirely adequate.

5.1.2 Radiochemical Pilot Plant Operations, Building 3019

Health Physics monitoring and surveillance efforts were continued for operations in support of the Light Water Breeder Reactor (LWBR) program. Radiochemical cell operations included: dissolution of ~ 218 kg of ^{233}U and $\sim 7.7 \times 10^3$ kg of thorium; solvent extraction purification of ~ 417 kg of ^{233}U ; and ion exchange purification of ~ 468 kg of ^{233}U . The oxide Conversion System (gloved box contained) produced ~ 496 kg of $^{233}\text{UO}_2$ which was packaged and monitored for off-site shipment.

One hundred forty-nine Radiation Work Permits were completed for the more hazardous operations. These included: modification and maintenance on the cell process equipment and sampling system; replacement and cleaning of the hydrogen furnace exhaust venturis; replacement of the disolver charging box window; replacement of various gloved box and process exhaust HEPA filters; modifications to the process off-gas system; decontamination and equipment removal in the ^{238}Pu and ^{233}U contaminated cubicles of Cell 4; and partial decontamination of ion exchange and solvent extraction process equipment and cells at the end of the LWBR support program.

Bio-assay sample data and Whole Body Counter data on personnel involved in the above operations indicated no significant internal exposure. The maximum annual whole body or critical organ dose equivalent to personnel from external sources (principally from ^{232}U daughter activities) was 2.24 rem.

The several releases of radioactive material which did occur were of minor consequence and were confined to established zones where detection and control procedures were effective.

5.1.3 Analytical Chemistry Operations in Buildings 3019 and 2026

Health Physics and Safety Surveys personnel provided consultation and monitoring services for Analytical Chemistry Division operations involving a broad spectrum of radionuclides. Radiation fields to which personnel were exposed and spread of radioactive contamination were minimized through the prompt response of operating personnel to any problem areas detected through health physics surveillance activities. Thirty-three Radiation Work Permits were completed for operations with the high risk potentials. These included: the renovation of HRLAF Cell 3 and installation of a lathe for machining samples from Peach Bottom Reactor graphite pieces; decontamination and relamping of HRLAL Cells 2 and 6; replacement of the HRLAL vacuum system blower and motor; removal and packaging of HRLAF and HRLAL cell wastes for disposition in solid waste storage facilities; replacement of HRLAF cell exhaust HEPA filters; decontamination of HRLAF cell 6; and replacement of the emission spectrograph arc furnace and furnace exhaust HEPA filter.

5.1.4 Hot Cell Operations, Buildings 3026-D and 3525

The work level of the Hot Cell Operations Group at the High Level Segmenting Cell Facility (Bldg. 3026-D) and the High Radiation Level Examination Laboratory (Bldg. 3525) continues apace. Most of their efforts are expended on samples or experiments which are intensely radioactive by virtue of having been irradiated in various reactors around the country. The demand for these services can be illustrated by pointing out that over 600 shipments (in and out) of shielded carriers in varying sizes were handled during the year. Additionally, over 40 loads of waste material were generated and transferred to the Solid Waste Facility for disposition.

Each of these operations required careful monitoring by Radiation and Safety Surveys personnel in order to assure that radiation shielding and contamination containment provisions conform to Laboratory standards. The maximum personnel exposures was < 25% of the permissible level, and contamination problems were confined to zoned areas where controls were adequate.

5.1.5 Laboratory Decontamination, Building 3028

After some years of operation as C-Zones because of contamination, the laboratories on three floors of Building 3028 were cleaned up and made available for other programs. The second and third floor levels had been contaminated with ^{244}Cm (up to $5 \times 10^4 \text{ } \mu\text{Ci/m}^2$), and the fourth level had a history of ^{147}Pm (up to 20 mrad/hr). After some eight weeks of cleaning, satisfactory decontamination was achieved and all surfaces were painted.

5.1.6 High Level Chemical Development Facility, Building 4507

The studies planned for LWR and LMFBF programs required complete renovation of Cells 3 and 4, which had been used as a Curium Recovery Facility. Interior cell surfaces and equipment were grossly contaminated with transuranium isotopes and mixed fission product activities. The general radiation background was ~ 300 mrem/hr, and transferable alpha contamination levels exceeded 2×10^6 d/m. Limited working times and positive air supplied plastic suits were necessary for cell entries by personnel.

Following equipment removal, partial decontamination was achieved by pressure spraying and sand blasting. Two coats of epoxy paint were then applied to fix the residual activity in place. At the end of the year, Cell 3 was ready for the fuel study program and Cell 4 was ready for equipment installation.

5.1.7 Transuranium Research Laboratory, Building 5505

Health Physics and Safety personnel assigned to the TRL provided continuing assistance in the form of radiological and safety surveillance in the design and operation of facilities and equipment for research programs to investigate the chemical, physical, and nuclear properties of a number of transuranic isotopes. Design developments during the year included: 1) an inert atmosphere glove box complex for the preparation and study of milligram quantities of transplutonium metals, 2) a solution microcalorimeter facility, 3) a magnetic susceptibility facility, 4) a de-mister/pressure control system for contaminated exhausts from glove box vacuum pumps, and 5) an improved solid state alpha survey instrument for use in areas where neutron radiation backgrounds exist.

One staff member presented a paper entitled, "Radiological Safety Considerations in the Design and Operation of the ORNL Transuranium Research Laboratory" at an IAEA symposium in Otaniemi, Finland.

5.1.8 Transuranium Processing Activities, Buildings 7920 & 7930

Radiation and Safety Surveys personnel provided the surveillance necessary for the continuing operations in these buildings: the processing of the intensely radioactive target rods after irradiation in the High Flux Isotope Reactor; the fabrication and shipment of neutron sources and other research materials (from ^{248}Cm up to ^{254}Es); maintenance operations in all pits and cubicles (alpha contaminated in the range up to 10^7 d/m/cm²); and waste removal where spontaneous fission isotope contamination generated neutron exposure dose rates up to 2 rem/hour. Careful planning, diligent execution, and detailed monitoring are essential to the safe handling of such high hazard index, high specific activity materials. Personnel exposure dose equivalents were maintained well below permissible values, and contamination was confined to the zoned areas where controls were adequate.

5.1.9 Coolant Salt Technology Facility, Building 9201-3

Tritium behavior studies in this facility were completed and the loop was placed in stand-by condition. Some 140 Ci of tritium had been used in experiments over a one-year period. Although major components and piping were left in place for possible future work, the drain tank containing over a ton of tritium-contaminated sodium fluoroborate was disconnected, sealed, and transported to ORNL for disposition in the Solid Waste Storage Area. The operation was completed with no significant releases of contamination.

5.1.10 Calutron Tank Removal, Building 9204-3

A complete calutron installation was removed as a unit from the alpha containment area, packaged, and transported to ORNL for disposition in the Solid Waste Storage Area. The system weighed ~ 10 tons and was estimated to contain up to 1 gram of Curium isotopes (~ 85 curies). The only adverse aspect of the operation was a small spill of contaminated liquid which resulted in contamination of two employees. Bio-assay sampling showed their intakes to be < 25% of the permissible values.

5.1.11 Plutonium Experiment, Building 9210

Special facilities were installed to enable the performance of a specific-locus mutation experiment involving Plutonium-239. Two hundred mice were each injected with 0.25 μ Ci. The animals were then placed in isolation chambers under controlled ventilation conditions. Health physics surveillance was provided at each step of the experiment: injection, handling, cage cleaning, etc., through to animal sacrifice and waste disposal. Careful planning and detailed monitoring resulted in an incident-free experiment with highly hazardous material.

5.1.12 Shale Fracturing Facility

Continuous monitoring was provided for Halliburton Company personnel (under contract with ORNL) during major repairs on the Shale Fracturing injection pump. Through special shielding, some decontamination, and careful planning, exposures and contamination were adequately controlled.

5.2 X-Ray Safety Program

During 1976, the x-ray safety program emphasized radiation safety around analytical x-ray devices. X-ray machines of this type have, in the past, been involved in a sizeable fraction of the accidental exposures reported by ERDA and the State Public Health Department.

The program initiated the previous year for updating the safety systems on the most hazardous of the x-ray diffraction and x-ray fluorescence units was completed. The program was mainly directed toward preventing accidental exposures to individuals by the intense direct

beam used in analytical work. In most cases interlock switches were added which would turn the x-ray unit off when a piece of equipment is removed which exposed the direct beam. Fixed beam barriers were installed near the beam ports on a couple of units and an interlocked fence was added at one installation.

A film, "The Double-Edged Sword", was shown to several groups. This film stressed radiation safety at analytical x-ray installations.

A fail-safe warning light system was added to an old x-ray fluoroscope cabinet unit.

During the year several lectures on x-ray safety were given to I&C and Health Physics and Safety Survey personnel.

5.2.1 Microwave Safety Program

The number of microwave ovens has been steadily increasing. At the end of the year there were 21 ovens. All microwave ovens were surveyed quarterly for microwave leakage and interlock integrity. Leakage on all units was found to be within federal limits. One unit which receives extra heavy use was found to have slightly elevated leakage due to deteriorating door seals.

A microwave plasma unit which can produce microwave fields in occupied areas on the order of magnitude of the federal limits has been surveyed every two to three weeks.

Classes on microwave safety were given to Health Physics and Safety Survey personnel and to ORAU trainees.

5.3 Radiation Incidents

The Applied Health Physics Annual Report for 1959 (ORNL-3073) described a system for classifying radiation accidents, or near accidents, and the term "unusual occurrence" was adopted to describe these events. ERDA has recently selected the term "unusual occurrence" to cover a much wider range of activities than the initial definition in the Applied Health Physics Annual Report. Subsequently, Applied Health Physics and Safety has adopted the term "radiation incidents" to describe operational incidents involving ionizing radiation and this term is defined in Procedure 2.6, Reporting of Radiation Incidents, Procedures and Practices for Radiation Protection, (ORNL Health Physics Manual), as follows:

- a. A Radiation Protection Standard for External or Internal Exposure is exceeded.
- b. An incident involving an unplanned personnel exposure when the external exposure for one day exceeds (a) 0.1 rem for exposure to the total body, head and trunk, lens of the eye, gonads, or blood forming organs, (b) 0.3 rem for exposure to the skin of the body, and (c) 1.5 rem for exposure to the extremities.

- c. An incident involving an internal exposure such that the first sample following the event exceeds the Excretion Index (E.I.) and/or the first two samples following the incident exceed 25% of the E.I.
- d. An incident involving the uncontrolled release of radioactivity which results in a significant interruption of normal operations.
- e. An incident resulting in concentration levels above the concentration guides in an unzoned area, when averaged over 24 hours.
- f. An incident resulting in recovery, evaluation, or decontamination cost which exceeds \$1,000.
- g. An incident where one or more employees express "undue" concern about their radiation exposure.
- h. A willful and/or repeated violation of a Health Physics procedure.
- i. An incident which has an adverse effect on public relations.
- j. An incident in which internal monitoring or the expeditious processing of external radiation monitoring devices is requested for a non-employee.

Due to a change in definitions, it is not practical to compare the number of radiation incidents with the number of unusual occurrences reported in past years. Trends in number and types of incidents will be observed and reported over the next few years. There were 17 radiation incidents (unusual occurrences) reported in 1976. All of the radiation incidents were considered as minor events with the exception of one incident in which an employee received a dose to the skin of about 15 rem.

5.4 Laundry Monitoring

Approximately 537,000 articles of wearing apparel and 157,000 articles such as mops, bags, towels, etc. were monitored at the Laundry during 1976. Approximately five percent were found contaminated. Of 410,163 khaki garments monitored during the year, only 67 were found contaminated.

A total of 5,753 full face respirators and 4,999 canisters were monitored during the year. Of this number 443 required further decontamination after the first cleaning cycle.

6.0 INDUSTRIAL SAFETY AND SPECIAL PROJECTS

The safety record for 1976 surpasses the record of the previous year which was considered one of the best in the history of the Laboratory. Only one disabling injury occurred during the year, and ORNL employees worked a continuous period of 309 days without a disabling injury. For the second consecutive year the Laboratory earned the highest awards of both the Union Carbide Corporation and the National Safety Council: the Award of Distinguished Safety Performance and the Award of Honor. In addition ERDA's Award of Achievement was earned for a reduction of greater than 25% in the injury incidence rate compared with the 1975 record.

6.1 Accident Analysis

The disabling injury frequency rate for 1976 of 0.13 equals the best previous rate which was established in 1968. The frequency rate of 1.33 for recordable injuries and illnesses was the best in the Nuclear Division and showed an improvement of approximately 65% over last year's Laboratory record. The injury statistics for ORNL for the period 1960-1976 are shown in Table 6.1.1, page 56. The disabling injury history of ORNL for the past five years is shown in Table 6.1.2, page 57; and the disabling injury frequency rates since the inception of Carbide's contract as compared with NSC, ERDA, and UCC are shown in Table 6.1.3, page 58.

Eleven ORNL divisions did not have a recordable injury or illness in 1976. Injury statistics by divisions are shown in Table 6.1.4, page 59

Disabling injury accident-free periods for ORNL are shown in Table 6.1.5, page 60. From November 12, 1975, through September 15, 1976, the Laboratory accumulated over 6.3 million manhours without a disabling injury. These figures represent the most hours worked without a disabling injury since 1968-69 when the Laboratory accumulated an accident-free period of over 8.5 million manhours.

Table 6.1.6, Figure 6.1.1, and Table 6.1.7, pages 61, 62, and 63, present ORNL injury data according to type, part of body injured and nature of injury.

A tabulation of injuries for the four UCC-ND facilities is shown in Table 6.1.8, page 64. ORNL's frequency rate of 0.13 for disabling injuries and 1.33 for recordable injuries and illnesses was best within the Nuclear Division facilities.

Statistics on motor vehicle accidents, fires, and off-the-job disabling injuries are shown in Tables 6.1.9, 6.1.10 and 6.1.11, pages 65 and 66.

Off-the-job disabling injury frequency rate for ORNL employees was 2.9 compared with ORGDP's frequency rate of 3.58, Paducah's 5.44 and Y-12's rate of 5.08. The number of off-the-job fatalities during 1976 was abnormally high. Three of the five fatalities were the result of automobile accidents.

6.2 Summary of Disabling Injury

The following is a summary of the disabling injury that occurred in 1976: Date of Injury - September 16, 1976.

A lineman employee of the Plant and Equipment Division received serious electrical burns when accidentally exposed to an energized 13.8 kV electrical power line. The accident occurred while the lineman was installing insulating materials on the energized line. The initial current path to ground was apparently established by the lineman's right hand in proximity of the ground wire while he simultaneously contacted an exposed portion of the phase C conducting material with his left forearm. The lineman sustained burns over approximately 50% of his body including second and third degree burns on his chest, back and arms and second degree burns around his waist, neck and lower part of his face. Time loss estimated - 349 days.

6.3 Safety Awards

The excellent safety record established during 1976 earned for each Laboratory employee, at the X-10 site and on the payroll as of December 31, 1976, an \$18.00 value safety incentive award. The items available as an award will be contained in an award booklet from which the employee will have the opportunity to select from seventy or more merchandise items. The item chosen will be mailed directly to the home of the employee.

Table 6.1.1 ORNL Injury Statistics (1960-1976)

Year	Disabling Injuries			Serious Injuries	
	Number	Frequency	Severity	Number	Frequency
1960	6	0.94	77	99	15.5
1961	10	1.55	576	80	12.4
1962	10	1.45	377	70	10.2
1963	11	1.55	172	58	8.2
1964	8	1.07	148	83	11.1
1965	18	2.34	366	97	12.6
1966	5	0.64	155	93	11.9
1967	4	0.50	266	89	11.1
1968	1	0.13	8	73	9.4
1969	2	0.27	9	37	4.9
1970	5	0.76	88	49	7.5
1971	4	0.61	298	38	5.8
1972	7	1.08	52	49	7.6
1973	2	0.33	24	35	5.8
1974	5	0.81	51	30	4.9
1975	2	0.27	24	82	2.25*
1976	1	0.13	14	51	1.33

* Since 1975, the serious injury frequency rate has been based on OSHA system for recording injuries and illnesses.

Table 6.1.2 Disabling Injury History--ORNL (1972-1976)

	1972	1973	1974	1975	1976
Number of Injuries	7	2	5	2	1
Labor Hours (Millions)	6.5	6.0	6.2	7.3	7.6
Frequency Rate	1.08	0.33	0.81	0.27	0.13
Days Lost or Charged	337	692	315	173	106
Severity Rate	52	115	51	24	14

Table 6.1.3 ORNL Disabling Injury Frequency Rates Since Inception
of Carbide Contract Compared with Frequency Rates
for NSC, ERDA and UCC

Year	ORNL	NSC	ERDA	UCC
1948	2.42	11.49	5.25	5.52
1949	1.54	10.14	5.35	4.91
1950	1.56	9.30	4.70	4.57
1951	2.09	9.06	3.75	4.61
1952	1.39	8.40	2.70	4.37
1953	1.43	7.44	3.20	3.61
1954	0.79	7.22	2.75	3.02
1955	0.59	6.96	2.10	2.60
1956	0.55	6.38	2.70	2.27
1957	1.05	6.27	1.95	2.41
1958	1.00	6.17	2.20	2.21
1959	1.44	6.47	2.15	2.16
1960	0.94	6.04	1.80	1.92
1961	1.55	5.99	2.05	2.03
1962	1.45	6.19	2.00	2.28
1963	1.55	6.12	1.60	2.10
1964	1.07	6.45	2.05	2.20
1965	2.34	6.53	1.80	2.40
1966	0.64	6.91	1.75	2.57
1967	0.50	7.22	1.55	2.06
1968	0.13	7.35	1.27	2.24
1969	0.27	8.08	1.52	2.49
1970	0.76	8.87	1.28	2.27
1971	0.61	9.37	1.44	2.05
1972	1.08	10.17	1.40	1.73
1973	0.33	10.55	1.45	1.50
1974	0.81	10.20	1.60	0.99
1975	0.27	13.10	2.50	0.61
1976	0.13	-----	-----	0.86

Table 6.1.4 Injury Statistics by Division--1976

Division	Medical Reports Received	Recordable Injuries and Illnesses		Disabling Injuries			Exposure Hours (In Millions)
		Number	Frequency	Number	Frequency	Severity	
Analytical Chemistry	12	1	0.85				.236
Chemical Technology	26	1	0.37				.537
Chemistry	5	0	0				.197
Central Management	1	0	0				.093
Physics	6	0	0				.195
Instr. and Controls	21	2	0.77				.517
Health Physics	13	1	0.63				.316
Metals and Ceramics	18	0	0				.552
Neutron Physics	3	0	0				.142
Computer Sciences	12	2	1.10				.363
Solid State	2	0	0				.138
Engineering	9	0	0				.447
Health	1	0	0				.069
Inspection Engineering	2	0	0				.068
Laboratory Protection	10	2	1.93				.207
Operations	41	4	1.52				.528
Employee Relations	12	2	1.84				.217
Plant and Equipment	276	32	3.85	1	0.60	64	1.661
Information	9	0	0				.458
Environmental Sciences	11	1	0.74				.272
Energy	1	0	0				.126
Finance and Materials	17	3	1.97				.304
PLANT TOTAL	508	51	1.33	1	0.13	14	7.644

Table 6.1.5 Disabling Injury Accident - Free Periods--ORNL (1972-1976)

Accident-Free Period	Man-Hours Accumulated
September 27, 1971 - January 23, 1972	2,021,680
January 25, 1972 - April 10, 1972	1,396,282
April 12, 1972 - June 20, 1972	1,262,911
June 22, 1972 - June 29, 1972	151,434
July 1, 1972 - October 15, 1972	1,874,592
October 17, 1972 - November 19, 1972	630,669
November 21, 1972 - December 10, 1972	296,276
December 12, 1972 - April 25, 1973	2,327,051
April 27, 1973 - July 29, 1973	1,428,975
July 31, 1973 - January 15, 1974	2,760,549
January 17, 1974 - May 6, 1974	1,869,338
May 8, 1974 - June 15, 1974	661,399
June 17, 1974 - August 11, 1974	926,437
August 13, 1974 - December 5, 1974	2,010,547
December 7, 1974 - April 6, 1975	2,570,944
April 8, 1975 - November 10, 1975	4,543,462
November 12, 1975 - September 15, 1976	6,375,994
<u>Best Accident-Free-Period</u>	
July 4, 1968 - August 20, 1969	8,529,750

Table 6.1.6 Number and Percent of Accidents by Type - 1976

Type of Accident	Number	Percent
Struck Against	233	47.7
Struck By	105	21.5
Slip, Twist	32	6.6
Caught In, On, Between	42	8.6
Contact with Temp. Extremes	27	5.5
Fall, Same Level	26	5.3
Inhalation, Absp., Ingestion	4	0.9
Fall, Different Level	10	2.1
Other	<u>9</u>	<u>1.8</u>
TOTAL	488	100.0

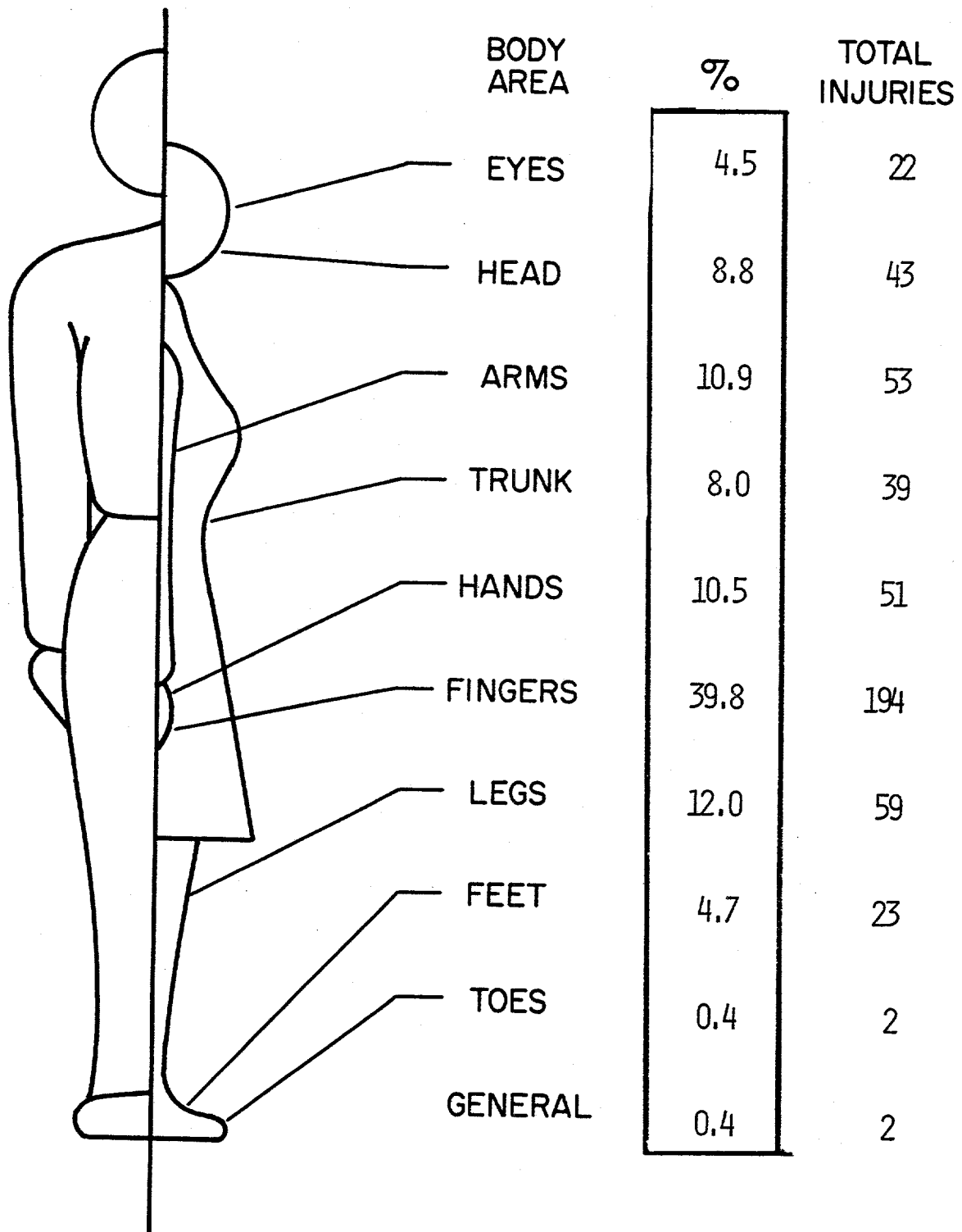


Fig. 6.1.1 Part of Body Injured

Table 6.1.7 Number and Percent of Accidents
by Nature of Injury - 1976

Nature of Injury	Number	Percent
Laceration, Puncture	232	47.5
Contusion, Abrasion	126	25.8
Strain	34	7.0
Burn, Temperature	32	6.6
Sprain	10	2.0
Conjunctivitis	21	4.4
Burn, Chemical	10	2.0
Other	<u>23</u>	<u>4.7</u>
TOTAL	488	100.0

Table 6.1.8 'Tabulation of Injuries by UCC-ND Facility--1976

Plant	Labor Hours (Millions)	Disabling			Recordable Injuries and Illnesses		
		Number of Injuries	Frequency Rate	Days Lost or Charged	Severity Rate	Number of Injuries*	Frequency Rate
ORNL	7.6	1	0.13	106	14	51	1.33
ORGDP	10.8	4	0.37	454	42	94	1.74
Y-12	10.7	4	0.37	241	23	76	1.42
Paducah	4.1	4	0.97	458	111	71	3.43

* Includes the number of Disabling Injuries.

Table 6.1.9 Motor Vehicle Accidents (1972-1976)

Year	Number	Frequency	Damage
1972	12	5.93	\$4641
1973	10	5.22	\$ 915
1974	15	8.14	\$1968
1975	7	3.33	\$2567
1976	14	6.42	\$5136

Table 6.1.10 Number of Fires (1972-1976)

Year	Number	Damage
1972	23	\$ 0
1973	20	\$ 300
1974	8	\$ 0
1975	8	\$16,493
1976	0	\$ 0

Table 6.1.11 Number and Type of Off-The-Job
Disabling Injuries (1972-1976)

	1972	1973	1974	1975	1976
Transportation	3	5	8	14	20
Home	11	3	17	16	17
Public	3	5	10	6	9
Total	17	13	35	36	46
Days Lost	990	612	1197	1724	1251
Frequency	1.25	1.01	2.54	2.33	2.91
Fatalities	0	1	2	1	5

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